Using Code RELAP5 and RELAP/SCADAPSIM for Research Reactor LVR-15 and Experimental Equipments

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ABSTRACT

This paper describes spheres of using code RELAP5/3.2.2 and RELAP/SCADAPSIM by Reactor Services Division of Nuclear Research Institute in the Czech Republic. RELAP5/3.2.2 and RELAP/SCADAPSIM are used for the assessment of transient analysis of research reactor LVR-15 and for simulation of thermal-hydraulic conditions of high-pressure and high-temperature in-pile water loop facilities for testing of materials under various conditions of LWRs. Recently analysis are also made for the in-pile helium loop facility build with the aim to simulate thermal-hydraulic conditions and structure materials behaviour under VHTR conditions.

1 INTRODUCTION

The research reactor LVR-15 is situated at Nuclear Research Institute (NRI) in Rez, Czech Republic, and is operated by Reactor Services Division. There is a number of experimental equipments for irradiation and testing of structural materials under typical PWR and BWR thermal-hydraulic and water chemistry conditions. For example, the in-pile loop RVS-3 is currently used for the study of corrosion behaviour of advanced fuel cladding materials under PWR conditions employing electrically heated fuel rods imitators. Recently NRI Rez participates in the EU program focused on development of VHTR technologies - RAPHAEL / SP-ML (Materials Development). The NRI is taking part in development and construction of dedicated experimental facility (an in-pile helium loop), which will be able to ensure experimental conditions of the VHTR concept.

2 RELAP5/MOD3.2.2 AND RELAP/SCADAPSIM

2.1 Code RELAP5/MOD3.2.2

The RELAP5[1] has been developed at Idaho National Engineering and Environmental Laboratory (INEEL) for the U.S. Nuclear Regulatory Commission. The RELAP5 code has
been developed for best-estimate transient simulations of a light water reactor coolant system during postulated accidents (transients and loss-of-coolant-accident). The RELAP5 is highly generic code that can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear system involving the two-phase fluid.

### 2.2 RELAP/SCDAPSIM

RELAP/SCDAPSIM[2,3], designed to predict the behavior of reactor systems during normal and accident conditions, is being developed at Innovative Systems Software (ISS) as part of the international SCDAP Development and Training Program (SDTP). RELAP/SCDAPSIM uses the publicly available SCDAP/RELAP5 models developed by the US Nuclear Regulatory Commission in combination with proprietary (a) advanced programming and numerical methods, (b) user options, and (c) models developed by ISS and other members of the SDTP. RELAP/SCDAPSIM/MOD3.4 includes (a) new models for fission product transport and deposition, fuel assembly behavior, and in-vessel melt retention, (b) advanced programming and numerical techniques, and (c) integrated graphics displays.

### 3 SIMULATION OF THERMO-HYDRAULIC CONDITIONS IN RESEARCH REACTOR LVR-15

#### 3.1 Research reactor LVR-15 design

The research reactor LVR-15 is situated at the Nuclear Research Institute Rez, Czech Republic. The LVR-15 is a tank type experimental reactor moderated and cooled by demineralized water under atmospheric pressure with an average coolant temperature in the core of 320 K. The reactor design is shown in Fig.1 with a schematic of the coolant loop in Fig.2. Thermal power of the reactor is 10 MW. Currently, the core of LVR-15 reactor is loaded with IRT-2M type fuel elements with 36% enriched of $^{235}$U. The core of the reactor includes from 24 to 32 fuel assemblies and up to four irradiation positions are dedicated for the experimental irradiation facilities. The core configuration can be adapted according to the position of irradiation rigs. The fuel assembly consists of four or three square concentric tubes (Fig.3) where each tube is a sandwich type fuel element with thickness of 2mm composed of three layers. A central layer of 0.675mm containing UO$_2$ dispersed in Al matrix is covered from the both sides by Al alloy cladding (thickness of 0.65mm each).

The heat generated in the core is removed by demineralized water circulating in the closed primary circuit which contains two heat exchangers where the heat is transferred into the in the secondary circuit. Five circulation pumps maintain circulation of water in the primary circuit. The coolant in the reactor core flows downward during normal operation. Four of the five circulation pumps are mutually interchangeable and work as the main circulation pumps. The fifth pump is a part of the emergency cooling system. The schematic of the primary system is shown in Fig.2.

Summary of the irradiation conditions in reactor LVR-15:
- Fast neutron flux (E>0.1 MeV) - $1.2 \times 10^{18} - 3.2 \times 10^{18}$ (m$^{-2}$.s$^{-1}$)
- Thermal neutron flux - $1.4 \times 10^{18} - 2.2 \times 10^{18}$ (m$^{-2}$.s$^{-1}$)
- Gamma heating: 2-5 W/g
Figure 1. LVR15 reactor design

Figure 2. LVR-15 primary coolant flow circuit
3.2 Thermal-hydraulic analysis of reactor LVR-15 by code RELAP5/MOD3.2.2 and RELAP/SCADAPSIM

The developed RELAP model simulating the thermal-hydraulic conditions of the research reactor LVR-15 has been used for safety analysis and preparation of updated safety report of the reactor LVR-15. This model was tested and qualified by NRI and SDTP. (The RELAP/SCDAPSIM interactive 3D displays have been very helpful to verify that the input model is physically consistent with the reactor system). The nodalization scheme of the research reactor is shown in Fig.4. (Note: this nodalization scheme does not include all component’s detailed scheme). In hydrodynamic description, the core region of the reactor vessel is represented by 20 parallel channels (hot and two average fuel assemblies) together with a channel to model bypass. All core heat structures are modeled as cylindrical structures with total length equal to sum of all corresponding fuel rods. The core heat structures are divided in axial direction into 6 parts.

A number of calculations for reactivity changes and flow pumps fails (in the loss of coolant accident, simultaneous failure of all reactor coolant pumps is assumed) were performed. For example, an analysis of a transient involving the loss of all main circulation pumps shows that use of the one emergency pump is able to stabilize the reactor coolant flows quickly and eliminate any significant temperature excursions. In this transient, the reactor protection system scrams the reactor immediately (within 1 s) following the loss of the main circulation pumps with the emergency pump starting within 10 s. Typical results from modeling of the reactor transients are presented in Fig.5 where an evolution of the fuel cladding temperature is shown for the above mentioned transient.
Figure 4. RELAP5 nodalization of the LVR-15 reactor

Figure 5. Hot assembly outside wall temperature versus time after the beginning of transient

4 SIMULATION OF THERMAL-HYDRAULIC CONDITIONS IN THE IN-PILE WATER LOOP RVS-3

4.1 Description of RVS-3 loop

The RVS-3 loop is a closed stainless steel high-pressure, high-temperature system with an irradiation channel situated in the reactor core. The loop consists of a main circulation pump, heat exchangers, a pressurizer, a volume compensator, irradiation and testing channels, make-up water and sampling systems. The schematic of the RVS-3 loop together with the main characteristics is shown in Fig.6.
The loop is used for the investigation of simultaneous effect of the thermal-hydraulic, water chemistry and irradiation parameters on behaviour of various structure materials of the LWR primary systems. The performed research has been mainly focused on study of environmental degradation processes (e.g. stress corrosion cracking under irradiation), nuclear fuel cladding-water coolant interaction (e.g. corrosion of advanced zirconium alloys with without presence of radiation), an effect of water chemistry on deposition of corrosion products etc.

4.2 Thermal-hydraulic analysis of the RVS-3 loop

The RELAP5 code has been used to model thermal-hydraulic parameters in the both test channels - in the in-pile and out-of-pile channels. The input model for experimental loop RVS-3 was created for simulating and assessment of the steady-state and transients. The primary system was modeled by 48 volumes, 20 junctions and more than 30 heat structures [4]. The secondary system was simulated only partly using time dependent volumes. This model was tested and qualified by SDTP. (The RELAP/SCDAPSIM interactive 3D displays have been very helpful to verify that the input model is physically consistent with the reactor system). The steady-state model of the loop has been validated against experimental data and is used for detail determination of the experimental conditions (e.g. the coolant/surface temperature distribution, pressure, flow rate, void fraction etc.). Currently, an assessment of the accidental event is being performed (e.g. a leak from the primary piping – LOCA event).
5 SIMULATION OF THERMO-HYDRAULIC CONDITIONS IN HELIUM LOOP

5.1 Helium in-pile loop

The NRI is developing and preparing construction of the experimental facility, which will be able to ensure experimental conditions of the VHTR concept. The Helium in-pile loop will be situated in the LVR-15 reactor body with dedicated test (irradiation) channel located in the reactor core (occupying one core cell position of dimension 71.5x71.5 mm$^2$). Currently the design concept of the in-pile helium loop has been prepared. The layout of the Helium loop is shown in Fig. 7. The foreseen experimental conditions of the Helium in-pile loop are summarized in the following Table 1:

Table 1: Experimental conditions of the Helium in-pile loop

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>He temperature in the specimens region</td>
<td>900-1000°C</td>
</tr>
<tr>
<td>He pressure in the test section</td>
<td>~7 MPa</td>
</tr>
<tr>
<td>He coolant temperature range</td>
<td>500-1000°C</td>
</tr>
<tr>
<td>Temperature of the irradiation channel pressure tube</td>
<td>&lt;500°C</td>
</tr>
<tr>
<td>Inlet temperature of He circulator</td>
<td>500°C</td>
</tr>
<tr>
<td>Helium flow rate</td>
<td>0.05-0.1 kg/s</td>
</tr>
<tr>
<td>Fast neutron flux ($E&gt;0.1$ MeV) in the test section</td>
<td>$1.2-3.2\times10^{18}$ (m$^{-2}$.s$^{-1}$)</td>
</tr>
<tr>
<td>Thermal neutron flux in the test section</td>
<td>$1.4-2.2\times10^{18}$ (m$^{-2}$.s$^{-1}$)</td>
</tr>
<tr>
<td>Nuclear heating in the test section</td>
<td>2-5 W/g</td>
</tr>
</tbody>
</table>

![Design layout of the Helium loop](image_url)

Figure 7. Design layout of the Helium loop
5.2 Thermal-hydraulic analysis of the Helium in-pile loop

The input model for experimental Helium in-pile loop was created for simulating and assessment of steady-state and transients analysis. The primary system was modeled by 164 volumes, 18 junctions and more than 160 heat structures. The calculations have been performed by the code RELAP5 adapted for simulations of thermal processes when using helium coolant.

The main goal of the thermal-hydraulic analyses was to find out whether the proposed loop design approach will meet the experimental requirements. According to results of the thermal-hydraulic calculations the optimization of the construction of the loop have been proposed. As results of the performed analysis the following characteristics and parameters of the loop have been determined: helium coolant distribution in the loop system, helium pressure drop along the loop system, helium coolant mass flow rate, power of the electrical heater and the main characteristics of the regenerative heat exchanger. In order to reach the target temperature range in the loop test section a special ceramic layer on the central tube needs to be used. The effect of this ceramic layer (ceramic tube) on the helium coolant temperature distribution in the loop system is presented in Fig.8.

Figure 8. Comparison of temperature distribution in the helium loop
(Loop1 – system with the ceramic tube, Loop2 – system without the ceramic tube; He mass flow rate 0.067 kg/s, power of the electrical heater 49 kW)
6 CONCLUSIONS

The RELAP5 code is extensively used in NRI to model and assess thermal-hydraulic processes and condition in the research reactor LVR-15. The code has been used to perform necessary analysis of the steady-state conditions and selected transient events for the purpose of the reactor Safety Report.

Also, the RELAP5 code is used to model and determine thermal-hydraulic conditions in the experimental facilities. The results of the calculations are used in a different ways: (i) for the Safety Reports of the irradiation facilities to be presented to the Nuclear Regulatory Body, (ii) for precise determination of the experimental conditions and (iii) as a support of the design work in the case of the development of new irradiation equipments.

REFERENCES